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ATR LEU Fuel and Burnable Absorber Neutronics Performance Optimization by Fuel Meat Thickness Variation

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ABSTRACT

The Advanced Test Reactor (ATR) is a high power density and high neutron flux research reactor operating in the United States. Powered with highly enriched uranium (HEU), the ATR has a maximum thermal power rating of 250 MWth. Because of the large test volumes located in high flux areas, the ATR is an ideal candidate for assessing the feasibility of converting an HEU driven reactor to a low-enriched core. The present work investigates the necessary modifications and evaluates the subsequent operating effects of this conversion.

A detailed plate-by-plate MCNP ATR 1/8th core model[1] was developed and validated for a fuel cycle burnup comparison analysis. Using the current HEU U-235 enrichment of 93.0 % as a baseline, an analysis can be performed to determine the low-enriched uranium (LEU) density and U-235 enrichment required in the fuel meat to yield an equivalent K-eff between the HEU core and the LEU core versus effective full power days (EFPD). The MCNP ATR 1/8th core model will be used to optimize the U-235 loading in the LEU core, such that the differences in K-eff and heat flux profile between the HEU and LEU core can be minimized.

The depletion methodology MCWO was used to calculate K-eff versus EFPDs in this paper. The MCWO-calculated results for the LEU cases with foil (U-10Mo) types demonstrated adequate excess reactivity such that the K-eff versus EFPDs plot is similar to the reference ATR HEU case. Each HEU fuel element contains 19 fuel plates with a fuel meat thickness of 0.508 mm. In this work, the proposed LEU (U-10Mo) core conversion case with a nominal fuel meat thickness of 0.381 mm and the same U-235 enrichment (19.7 wt%) can be used to optimize the radial heat flux profile by varying the fuel meat thickness from 0.191 mm (7.5 mil) to 0.343 mm (13.5 mil) at the inner 4 fuel plates (1-4) and outer 4 fuel plates (16-19). In addition, 0.8g of a burnable absorber, Boron-10, was added in the inner and outer plates to reduce the initial excess reactivity, and the inner/outer heat flux more effectively. The optimized LEU relative radial fission heat flux profile is bounded by the reference ATR HEU case. However, to demonstrate that the LEU core fuel cycle performance can meet the Updated Final Safety Analysis Report (UFSAR) safety requirements, additional studies will be necessary to evaluate and compare safety parameters such as void reactivity and Doppler coefficients, control components worth (outer shim control cylinders, safety rods and regulating rod), and shutdown margins between the HEU and LEU cores.

1. Introduction

The Advanced Test Reactor (ATR) at the Idaho National Laboratory (INL) is a high power density and high neutron flux research reactor operating in the United States. Powered with highly enriched uranium (HEU), the ATR has a maximum thermal power rating of 250 MW_{th} with a maximum unperturbed thermal neutron flux rating of 1.0 x 10^{15} n/cm^2–s. The conversion of nuclear test reactors currently fueled with HEU to operate with low-enriched uranium (LEU) is being addressed by the reduced enrichment for research and test reactors (RERTR) program. The ATR is a representative candidate for assessing the necessary modifications and evaluating the subsequent operating effects encountered when converting from HEU to LEU.

The scope of this task is to assess the feasibility of converting the ATR HEU fuel to LEU fuel while retaining all key functional and safety characteristics of the reactor. Using the current HEU U-235 enrichment of 93.0 % as a baseline, the study will determine the LEU uranium density required in the fuel meat to yield an equivalent K-eff between the HEU core and LEU core after 125 effective full power days (EFPDs) of operation with a total core power of 115 MW. A lobe power of 23 MW is assumed for each of the five lobes. Then, the U-235 loading determined to yield an equivalent K-eff will be used to predict radial, axial, and azimuthal power distributions. The heat rate distributions will also be evaluated for this core and used to predict the core performance as related to the current Upgraded Final Safety Analysis Report (UFSAR) and the associated Technical Safety Requirements (TSR's).

2. Advanced Test Reactor Description

The ATR was originally commissioned in 1967 with the primary mission of materials and fuels testing for the United States Naval Reactors Program. The ATR is a high power density and high neutron flux research reactor with large test volumes in high flux regions. General characteristics for the ATR are given in the "Users Handbook for the Advanced Test Reactor." Powered with HEU, the ATR has a maximum thermal power rating of 250 MW_{th} with a maximum unperturbed thermal neutron flux rating of 1.0 x 10^{15} n/cm^2–s.

The ATR was designed to provide large-volume, high-flux test locations. The unique serpentine fuel arrangement provides nine high-intensity neutron flux traps and 68 additional irradiation positions inside the reactor core reflector tank, each of which can contain multiple experiments.

The ATR's unique control device design permits large power shifts among the nine flux traps. The ATR uses a combination of control cylinders or drums and neck shim rods. The control cylinders rotate hafnium plates toward and away from the core, and the shim rods, which withdraw vertically, are individually inserted or withdrawn to adjust power. Within bounds, the power level in each corner lobe of the reactor can be controlled independently.

The ATR has five lobes which are loosely coupled. These five lobes are identified as Northwest (NW), Northeast (NE), Center (C), Southwest (SW), and Southeast (SE). During full power operation, operators can maintain the desired lobe power by rotating the Outer Shim Control Cylinders (OSCC) and withdrawing/inserting the neck shim control rods. Each lobe can be viewed as a smaller, independent reactor, which means there are five reactors in the ATR. A Lobe-by-Lobe (LbyL) conversion strategy can be developed, minimizing the impacts to the important experiments within other lobes.

3. Detailed Plate-by-Plate MCNP ATR Full Core and 1/8th Core Models

The ability to accurately predict K-eff and fission power distribution within the 19 fuel plates using the MCNP model is essential to the ATR LEU core conversion design. The developed MCNP ATR full core plate-by-plate model is shown in Figure 1. The MCNP ATR full core model has been validated by comparing MCNP-calculated parameters with both data from the ATR Surveillance Data Acquisition System (ASUDAS),[2] and PDQWS[3]-calculated parameters using the validated ATR model. A detailed plate-by-plate MCNP ATR 1/8th core model as shown in Figure 2 was derived from the validated MCNP ATR full core model for the fuel cycle burnup analysis. This model is used to optimize the U-235 loading in the LEU core by minimizing the K-eff differences with respect to the HEU core after 125 EFPDs of operation at total core power of 115 MW (23 MW per lobe).

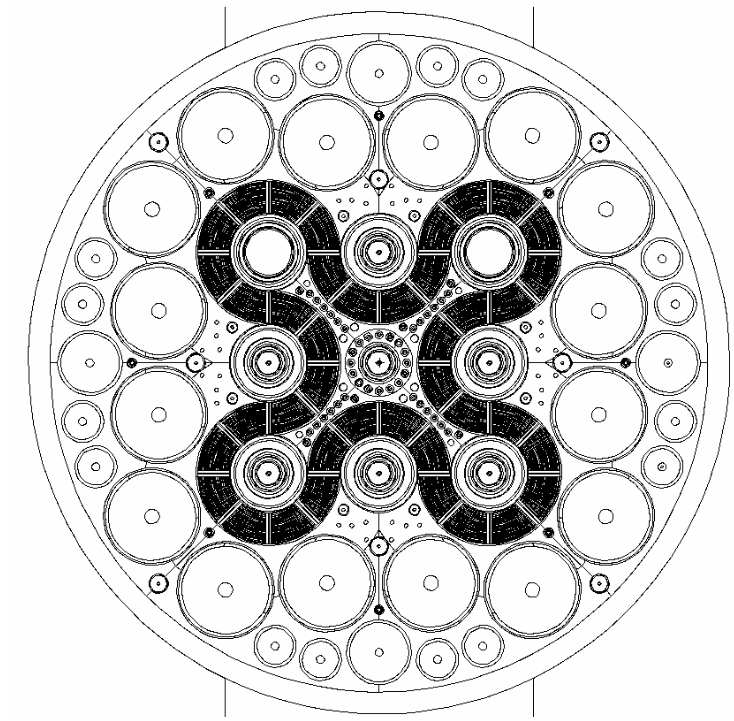

Figure 1. ATR MCNP full core model with 19 fuel plates per fuel element (FE).

4. MCWO – Fuel Burnup Analysis Tool

The fuel burnup analysis tool used in this study consists of a BASH script files that link together the two FORTRAN data processing programs, m2o.f[4] and o2m.f.[4] This burnup methodology couples the Monte Carlo transport code MCNP5[5,6] with the radioactive decay and burnup code ORIGEN2,[7] and is known as Monte Carlo with ORIGEN2, or MCWO.[4,8]

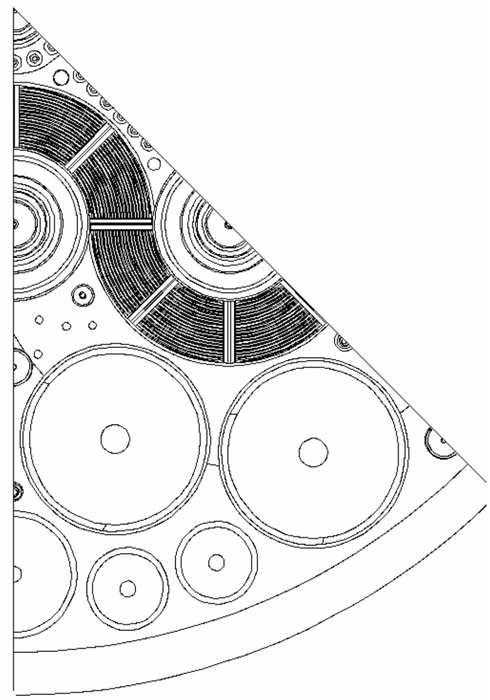

Figure 2. ATR SE-lobe 1/8th core MCNP model (FE 16-20).

The MCWO methodology produces criticality and burnup data based on various material feed/removal specifications, core power(s), and irradiation time intervals. MCWO processes user-specified input for geometry, initial material compositions, feed/removal specifications, and other problem-specific parameters. The MCWO methodology uses MCNP-calculated one-group microscopic cross sections and fluxes as input to a series of ORIGEN2 burnup calculations.

ORIGEN2 depletes/activates materials and generates isotopic compositions for subsequent MCNP calculations. MCWO performs one MCNP and one or more ORIGEN2 calculations for each user-specified time step. Due to the highly time-dependent nature of the physics parameters and material compositions of the modeled reactor system, the MCWO-calculated results are typically more accurate if long irradiation cycles are broken up into smaller intervals. It should be noted that an increase in the number of ORIGEN2 calculation steps does not significantly impact the overall MCWO execution time because MCNP dominates the MCWO execution time.

For each MCNP calculation step, MCNP updates the fission power distribution and burnup-dependent cross sections for each fuel plate then transfers data to ORIGEN2 for cell-wise depletion calculations. The MCNP-generated reaction rates are integrated over the continuous-energy nuclear data and the space within the region.

5. Neutronics Evaluation of HEU and Un-Optimized LEU

MCWO was used to perform an evaluation of the fuel cycle performance versus the EFPDs for: Case-A, ATR reference HEU, 20 mil thick fuel meat, 1075 g U-235, 0.66 g B-10 loading; Case-B1, Foil type LEU U-10Mo, 15 mil thick fuel meat with U-235 19.7 wt%, 1507.8 g U-235; and Case-C1, Foil type LEU U-10MO, 20 mil thick fuel meat with U-235 15.50 wt%, 1897.3 g U-235. Note Case-C1 was just one case in the overall feasibility study. However, the final enrichment will be 19.7 +-0.2 %. The analysis assumed that each nominal operating cycle was 50 EFPDs followed immediately by a seven day outage. Each 50 EFPD cycle was subdivided into 5 EFPD time step intervals. The outer shim control cylinder (OSCC) positions were set to 105°. The resultant MCNP-calculated tallies were normalized to a south lobe source power of 23 MW.

5.1 Comparison of K-eff Versus EFPDs

The MCWO-calculated K-eff for HEU Case-A, and LEU Case-B1 and -C1 are plotted in Figure 3. Please note that at the beginning of cycle (BOC) for each of the three nominal operating cycles modeled, the initial Xe poison was set to zero or decayed to a very small value during the 7 day shutdown time, thus causing a jump increase in K-eff.

The MCWO-calculated results of the bias adjusted K-eff versus EFPDs for Case-A demonstrates that the ATR HEU fuel provides adequate excess reactivity (for K-eff larger than one) for about 120 EFPDs of reactor power operation. The MCWO-calculated results of the bias adjusted K-eff versus EFPDs for Case-B1 and -C1 demonstrates that the LEU foil fuel types also provide much excess reactivity (for K-eff larger than one) for about 120 EFPDs of reactor power operation. The fuel densities for Case-B1 and -C1 were 16.88 g/cc. Because of no burnable B-10 and higher U-235 loading in the LEU cases, which cause the LEU have rather high K-eff at the beginning of life (BOL) and end of life (EOL). But, we need that excess reactivity to optimize the LEU radial heat flux profile by adding the burnable absorber B-10 and varying the thickness of the outer and inner 4 fuel meat as discussed in next section.

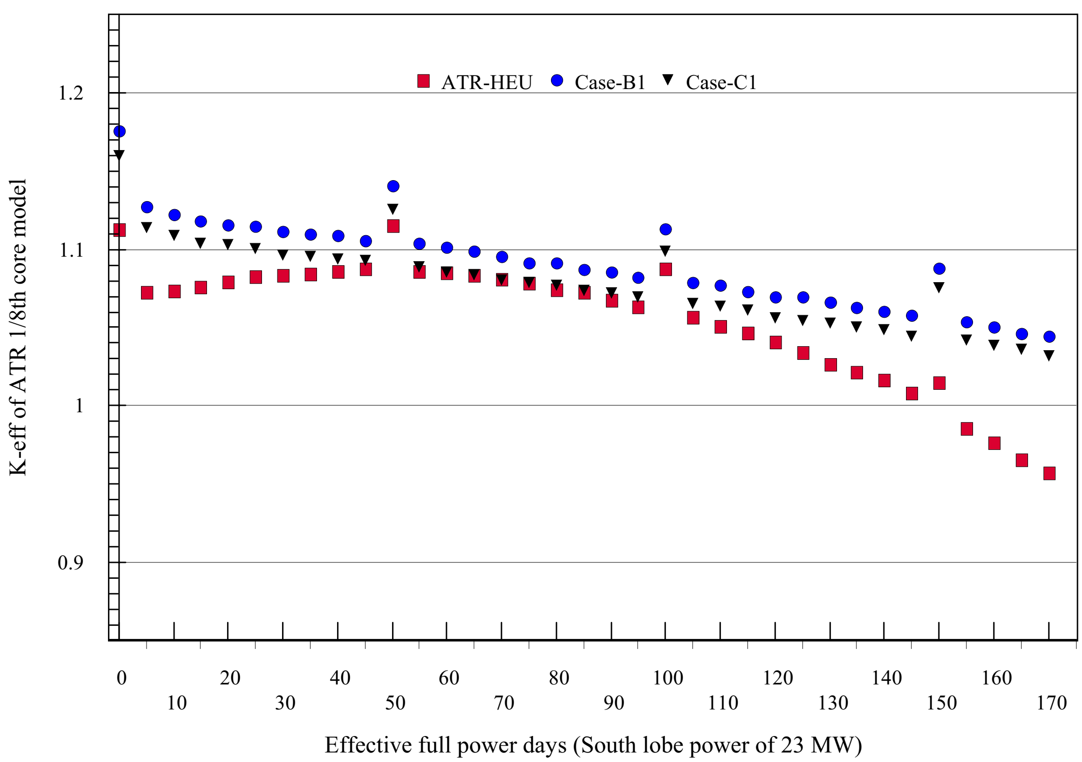

Figure 3. K-eff vs. EFPDs for ATR HEU Case-A, and un-optimized LEU Case-B1 and -C1.

5.2 Comparison of Radial Fission Power Profiles at BOC

For the beginning of the first cycle, the relative radial plate fission power heat flux was calculated using the MCWO methodology. Results for Case-A, -B1, and -C1 are plotted in Figure 4. It is apparent that when compared to Case-A, Case-B1 and -C1 yield significantly higher heat fluxes at the inner/outer plates.

In FE-18, the respective peak heat fluxes local-to-average-ratios (L2ARs) for Case-A, -B1, and -C1 were determined to be 1.22, 1.76, and 1.81, respectively. The peak flux occurred in plate 19 for all three cases. HEU Case-A has B-10 loading in the 4 inner/outer fuel plates (plates 1-4 and 16-19). The B-10 is a burnable poison which flattens the relative heat flux in the inner/outer plates to a peak value of about 1.22. Case-B1 and -C1 do not have any burnable absorber, therefore the peak relative heat flux ratio is approximately 1.8. From these results, it was established that the LEU fuels analyzed in Case-B1 and -C1 have rather high L2AR heat fluxes at both the inner/outer plates.

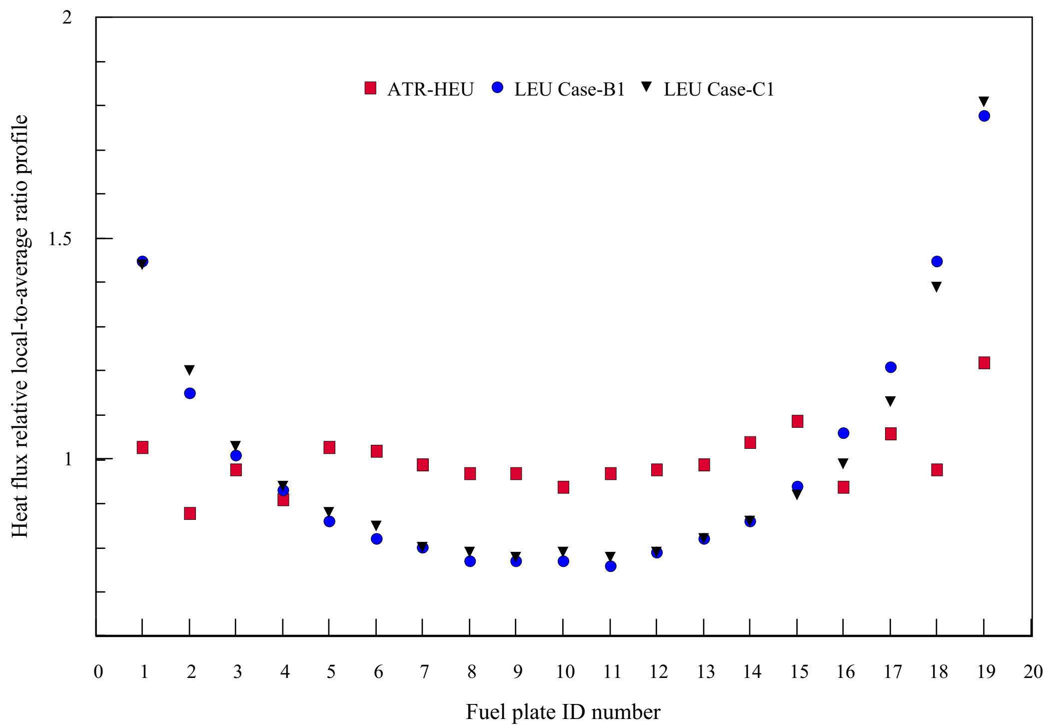

Figure 4. Radial fission power heat flux L2AR for ATR HEU Case-A, LEU Case-B1 and -C1.

For HEU reference Case-A, the lower L2AR at the inner/outer plates is due to the lower U-235 densities within those 4 inner/outer plates. For the HEU reference Case-A, the 4 inner/outer plates are loaded with 0.66 g of B-10, a burnable poison, which causes the heat flux profile to flatten even more when compared with LEU Cases. The HEU Case-A fuel plate specification and B-10 loading are given in Table 1.

To reduce the LEU heat flux L2AR, the U-235 contents and thickness of the inner/outer plates was evaluated and optimized. The LEU fuel loading was optimized such that the L2AR at the 4 inner/outer plates is bounded by reference HEU Case-A.

Table 1. Specifications for a standard ATR HEU FE with B-10 in the 4 inner/outer fuel plates.

HEU Plate	Plate Volume (cc)	U-235 Mass (g)	B-10 Mass (g)	U-235 Density (g/cc)
Plate-1	23.69	24.3	0.063	1.026
Plate-2	29.54	29.1	0.078	0.985
Plate-3	31.12	38.7	0.044	1.243
Plate-4	32.70	40.4	0.045	1.235
Plate-5	34.29	52.1	--	1.520
Plate-6	35.87	54.6	--	1.522
Plate-7	37.45	57.0	--	1.522
Plate-8	39.03	59.4	--	1.522
Plate-9	40.61	61.8	--	1.522
Plate-10	42.19	64.2	--	1.522
Plate-11	43.78	66.6	--	1.521
Plate-12	45.36	69.0	--	1.521
Plate-13	46.94	71.4	--	1.521
Plate-14	48.52	73.8	--	1.521
Plate-15	50.10	76.3	--	1.523
Plate-16	51.69	64.0	0.071	1.238
Plate-17	53.27	65.9	0.073	1.237
Plate-18	54.22	53.8	0.143	0.992
Plate-19	52.64	52.6	0.143	0.999
Total	792.99	1075	0.66	--

5.3 Azimuthal and Axial Fission Power Profiles

To investigate the azimuthal fission power L2AR profiles, plates 2-19 were subdivided into 10 azimuthal regions and plate 1 was subdivided into 8 azimuthal regions. To investigate the axial fission power L2AR profiles, the 48 inch fuel plate was axially subdivided into 32 equal regions.

The MCNP-calculated results indicate that all HEU and LEU cases have similar azimuthal and axial fission power profiles as shown in Figures 5 and 6. Therefore the average azimuthal and axial fission power profiles can and will be used for the fuel cycle burnup and thermal performance analysis.

6. Evaluation of HEU and Optimized LEU Fuel Cycle Performance

Based on the results of previously discussed comparisons, a study was performed to optimize the radial power profile of the LEU fuel plates such that the profile closely matches that of the HEU reference Case-A. The optimization was based upon a comparison of the calculated radial power profile for various LEU fuel loading schemes. The fuel loading schemes included varying the fuel meat thickness within the U10-Mo LEU fuel types.

6.1 Optimized LEU Radial Fission Power Profile at BOC

The optimization was achieved by varying the fuel meat thickness as well as loading the inner/outer plates with 0.8 g of B-10 as tabulated in Table 2. Table 2 summarizes the parameter variations that resulted in the flattest radial fission heat profile while still maintaining sufficient reactivity within the LEU core. To prevent the LEU fuel performance downgraded, the B-10 is loaded in the corresponding outer cladding.

Not surprisingly, the optimal LEU fuel loading is similar to the HEU reference case. The optimal LEU fuel loading has thinner fuel meat at the inner/outer plate positions. In summary; Case-B2, Foil type LEU U-10Mo, fuel meat with U-235 19.7 wt%, 1344.8 g U-235; and Case-C2, Foil type LEU U-10MO, fuel meat with U-235 15.50 wt%, 1694.8 g U-235. For the purposes of determining the feasibility of HEU to LEU conversion, the present study demonstrates a satisfactory loading scheme to achieve acceptable reactivity for three nominal 50 EFPD fuel cycles as well as maintain the radial heat flux L2AR profile.

Table 2. Fuel meat thickness variations used for radial power profile comparison studies.

Plate #	Case-B2 Fixed U-235 (wt%)	Case-B2 Varied fuel meat thickness (inches)	Case-C2 Fixed U-235 (wt%)	Case-C2 Varied fuel meat thickness (inches)	B-10 Mass (g) Total 0.8g
1	19.7	0.0090	15.50	0.012	0.063
2	19.7	0.0120	15.50	0.016	0.178
3	19.7	0.0135	15.50	0.018	0.044
4	19.7	0.0135	15.50	0.018	0.005
5	19.7	0.0150	15.50	0.020	0
6	19.7	0.0150	15.50	0.020	0
7	19.7	0.0150	15.50	0.020	0
8	19.7	0.0150	15.50	0.020	0
9	19.7	0.0150	15.50	0.020	0
10	19.7	0.0150	15.50	0.020	0
11	19.7	0.0150	15.50	0.020	0
12	19.7	0.0150	15.50	0.020	0
13	19.7	0.0150	15.50	0.020	0
14	19.7	0.0150	15.50	0.020	0
15	19.7	0.0150	15.50	0.020	0
16	19.7	0.0135	15.50	0.018	0.001
17	19.7	0.0120	15.50	0.016	0.033
18	19.7	0.0090	15.50	0.012	0.133
19	19.7	0.0075	15.50	0.010	0.343

The MCWO methodology was used to calculate the relative radial plate fission power heat flux for the optimized LEU cases for the beginning of the first cycle. In FE-18, the respective peak heat fluxes L2AR for Case-A2, -B2, and -C2 was determined to be 1.22, 1.13, and 1.15, respectively. Results for Case-A2, -B2, and -C2 are plotted in Figure 7. This plot demonstrates that Case-B2 and -C2 yield very similar radial L2AR profiles as compared to Case-A.

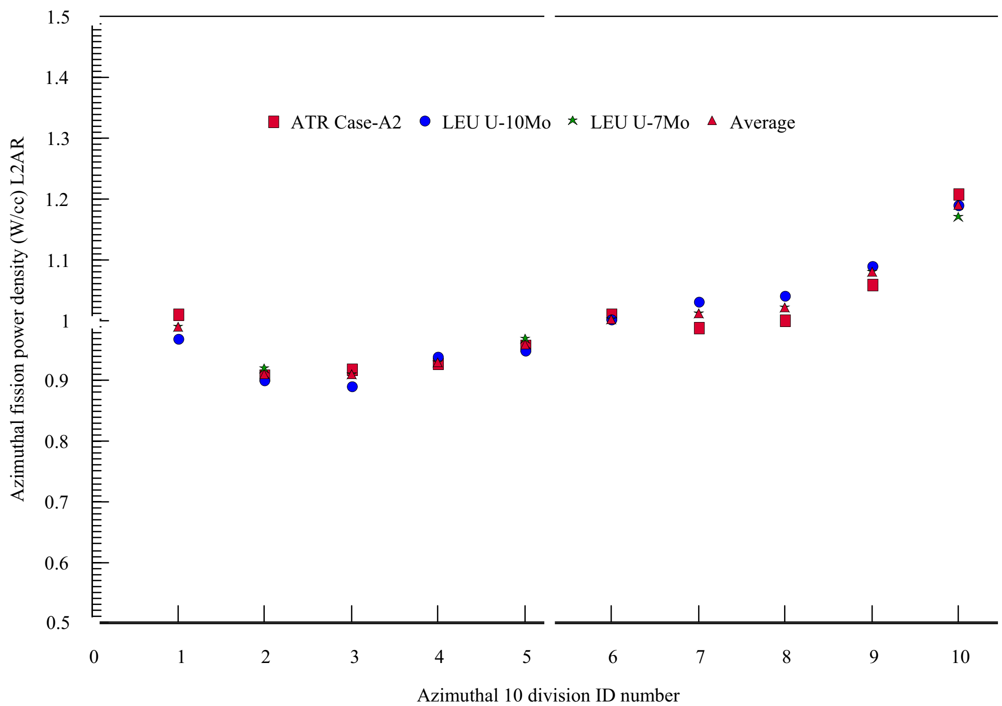

Figure 5. FE-18 Azimuthal distribution of fission power density L2AR.

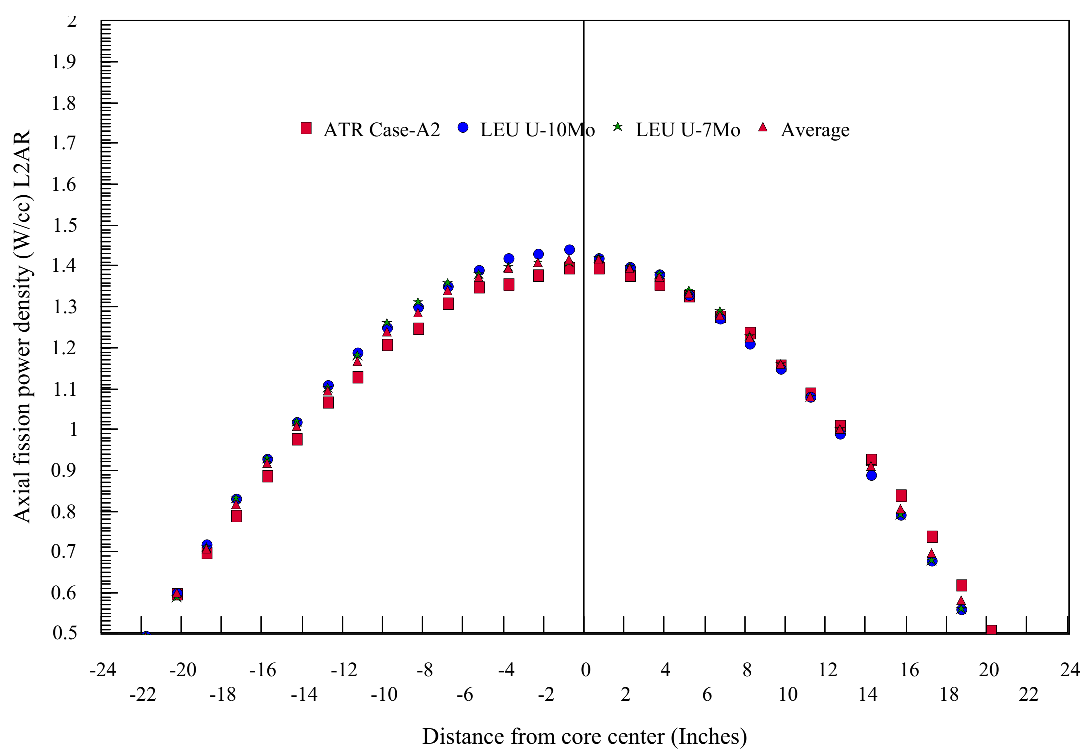

Figure 6. FE-18 Axial Distribution of fission power density L2AR.

6.2 Optimized LEU K-eff versus EFPDs

Using the optimized LEU fuel loadings, the MCWO-calculated K-eff for LEU Case-B2 and -C2 as a function of EFPDs as compared to the HEU reference Case-A is shown in **Figure** 8. Please note that the LEU fuels contain more U-238, which can be transmuted to Pu-239. Although the LEU cases have a lower K-eff at the BOC when compared with HEU Case-A, the LEU cases sustain operation for more EFPDs than HEU Case-A (at least 140 EFPDs).

7. Conclusions and Recommendations

For this study, the detailed plate-by-plate MCNP ATR 1/8th core model was developed and validated. This study also demonstrated that the 1/8th core model adequately represents the whole ATR core model for neutronics burnup analysis characterization. The detailed plate-by-plate MCNP ATR 1/8th core model used in this study handles complex spectral transitions at the boundaries between the plates in a straightforward manner.

The MCWO-calculated K-eff versus EFPDs results indicate that both LEU Case-B2 and –C2 provide excess reactivity versus burnup while providing fission heat profiles similar to HEU Case-A. The fixed U-235 enrichment (19.7 wt%) is the preferred approach to achieve the optimal fuel cycle performance. These studies indicate that the LEU radial L2AR profiles can achieve flattened profiles bounded by varying fuel meat thickness within the inner/outer 4 plates. The Case-B2 is selected for the safety analysis in the Final phase of the ATR LEU conversion feasibility study. As a result, it has been concluded that LEU core conversion for the ATR is feasible.

The LEU core designer can use the detailed plate-by-plate MCNP ATR 1/8th core model to optimize the U-235 loading by either minimizing K-eff differences with respect to the HEU core during the 125 EFPDs of operation at a total core power of 115 MW (23 MW per lobe), or by reducing the higher L2AR heat flux at the inner/outer plates. However, to demonstrate that the LEU core fuel cycle performance can meet the ATR UFSAR safety requirements, a further study will be necessary in order to investigate the detailed radial, axial, and azimuthal heat flux profile variations versus EFPDs.

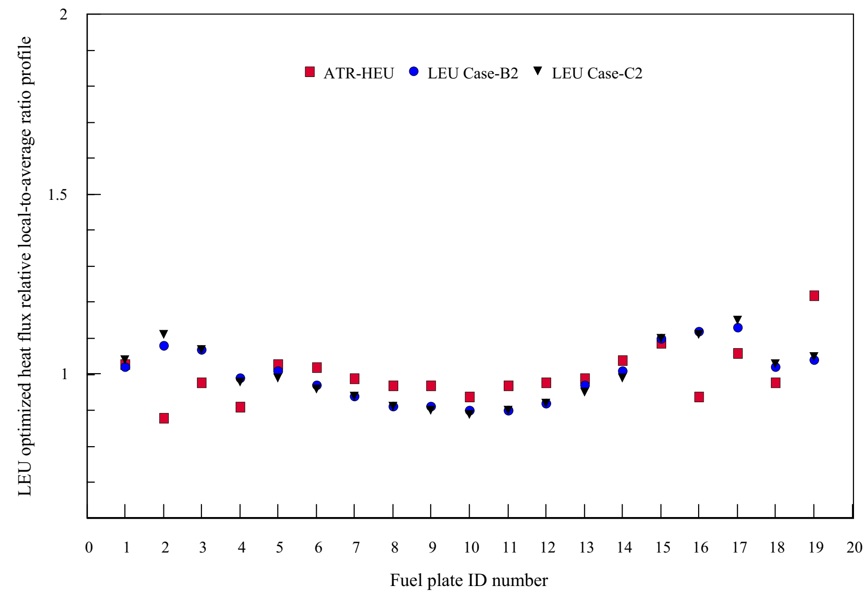

Figure 7. Fission power heat flux L2AR radial profiles for HEU Case-A2, LEU Case-B2 and -C2.

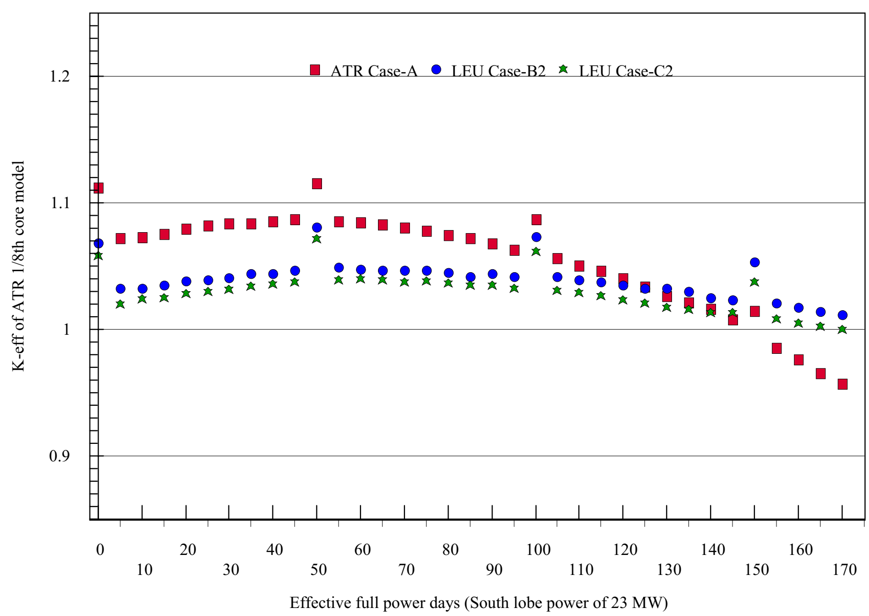

Figure 8. MCWO-calculated K-eff versus EFPDs for HEU Case-A, LEU Case-B2 and -C2.

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